US-APWRRAIsPEm Resource

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Subject: US-APWR Design Certification Application RAI 1055-7184 (1.9, 3.8, 5.1, 9.1, 12.3)
Attachments: US-APWR DC RAI 1055 RPAC 7184.pdf

MHI,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form on August 29, 2013 resulting in a clarification discussion on October 2, 2013. Your licensing review schedule assumes technically correct and complete responses within 60 days of receipt of RAIs.

Please submit your RAI response to the NRC Document Control Desk.

Thanks,

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REQUEST FOR ADDITIONAL INFORMATION 1055-7184

Issue Date: 10/07/2013

Application Title: US-APWR Design Certification - Docket Number 52-021

Operating Company: Mitsubishi Heavy Industries

Docket No. 52-021

Review Section: 09.01.02 - New and Spent Fuel Storage
Application Section: 1.9, 3.8, 5.1, 9.1, 12.3

QUESTIONS:

09.01.02-44

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. 10 CFR 52.47(a)(2) and 10 CFR 52.47(a)(5) require the applicant to describe and fuel handling systems and the kinds and quantities of radioactive materials produced in the facility. In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe the temporary fuel storage racks located in the Refueling Cavity, including the location, physical dimensions and elevations of the racks. The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to adding Figure 9.1.2-4 ,"Arrangement of the Containment Racks," and Figure 9.1.2-3 "Location of Containment Racks," Section View of Light Load Handling System," to the US-APWR DCD. The applicant's response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13012-P (R0) "Mechanical Analysis for US-APWR Containment Racks." MUAP-13012-P (R0) contains Figure 1-1 "Arrangement of Containment Racks (Plan View)" and Figure 1-2 "Arrangement of Containment Racks (Elevation View A-A)," which depict dimensional information about the temporary fuel storage racks, including elevation data. However, physical dimensions of the temporary fuel racks, including elevation data is not provided on Figure 9.1.4-2 and Figure 9.1.2-3 provided in the response to RAI 895-6172 Question 12.03-12.03-40 dated 25 April 2012.

Please revise and update the US-APWR DCD the drawings of the temporary fuel storage racks located in the Refueling Cavity, to include physical dimensions and elevations, or provide the specific alternative approaches used and the associated justification.

09.01.02-45

10 CFR 20.1101(b), 1201and 1202 require licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. In 10 CFR 20 the definition for ALARA includes guidance to make every reasonable effort to maintain exposures below regulatory limits, taking into account the state of technology. Regulatory Guide (RG) 1.206 section C.I.12.3.1 "Facility Design Features" notes that the Applicant should identify features that reduce the potential for exposure by minimizing the time in the area, reducing source build up, providing remote operation and reducing activation product generation. Regulatory Guide 8.8 Position C2.e, notes that the applicant should provide design features that reduce the potential for exposure by the selection of

materials to reduce activation product formation and finishing of the material surfaces for the purpose of minimizing erosion, facilitating decontamination and reducing deposition.

The applicant's response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included unsolicited proposed changes to US-APWR DCD pages 9.1-7, 9.1-9 and 9.1-11 which stated that "Surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel, and are smooth (changed from 125 AA to 250 micro inch) in accordance with the requirement of ANSI/ANS-57.2." However, the Electric Power Research Institute (EPRI) report TR-016780 "Advanced Light Water Reactor Utility Requirements Document" (URD), subsection 2.3.1.3.1.2 states "The refueling pool wall liner shall be surface finished to reduce the adherence of contamination and increase the efficiency of refueling pool decontamination activities after draining. The liner plate shall have a No. 4 surface finish or better and the liner plate welds shall be ground smooth." The reason given in the URD for this specification is that past LWR refueling experience has shown that one of the more time consuming critical path tasks is the decontamination of the refueling pool walls after refueling is completed. A smooth surface finish on the wall liners reduces the amount and depth of crevices which can accumulate contamination. The original US-APWR DCD roughness specification of 125 AA, corresponds to the No. 4 Finish which is approximately 36 to 60 micro inches.

Please revise and update the US-APWR DCD pages 9.1-7, 9.1-9 and 9.1-11 to provide surface finish specifications consistent with the 125AA value provided in the US-APWR DCD Revision 3, or provide the specific alternative approaches used and the associated justification.

09.01.02-46

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 63 "Monitoring fuel and waste storage," requires systems to ensure fuel safety. The guidance in RG-1.13 Revision 2 "Spent Fuel Storage Facility Design Basis," states that the minimum water depth above spent fuel should be 10 feet.

The applicant's response to RAI 524-4020 Revision 1, dated 14 September 2010; Question 12.03-12.03-35 Items 2 & 4 stated that MHI believes a rapid cavity drain down event is not considered feasible because the USAPWR permanent cavity seal (PCS) design prevents a seal cavity failure rapid drain down event and all cavity drain valves are administratively locked closed during fuel movement. In RAI 895-6172 Question 12.03-12.04-44 dated 27 January 2012 the staff asked the applicant to describe maximum feasible drain down rates and the associated dose rates from irradiated equipment or fuel. The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-44, dated 25 April 2012, stated that the drain down event assumed in the dose rate calculation is conservatively based upon the inadvertent opening of an 8-inch drain valve, and evaluates the worst case dose workers might receive during a refueling cavity drawdown of 5 feet during a 30-minute period (reasonable time for operators to identify the origin of the leak) before the leak is detected and mitigative actions are taken to close the valve and restore water level. The applicant continued to assert that higher drain down flow rates were not feasible. However, in addition to the pump assisted drain down flow rates already described by the staff in RAI 895-6172 Question 12.03-12.04-44, Institute of Nuclear Power Operations (INPO) Event Report Level 3 number 12-33 "Dislodged Steam Generator Nozzle Dam Bolting and Drain Plug During Eddy Current Inspections," described a near rapid drain down event due to steam generator nozzle dams bolts that were missing

causing the nozzle dam to bow out at the bottom because of the hydrostatic head pressure from the water in the refueling cavity.

Please revise and update the US-APWR DCD to describe drain down dose rate calculation assumptions that are consistent with assumptions used in US-APWR DCD Chapter 19 "Probabilistic Risk Assessment and Severe Accident Evaluation," and relevant industry operating experience, or provide the specific alternative approaches used and the associated justification.

09.01.02-47

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities", Part 68 “Criticality accident requirements,” paragraph (b)(6) states “Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.”

The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-42, dated 25 April 2012, stated that the containment racks were designed for all postulated normal and accident conditions and do not need criticality monitors. However, the radiation monitor requirement of 10 CFR 50.68(b)(6) is applicable regardless of the value of k-effective. The US-APWR DCD Revision 3 does not describe which radiation monitor fulfills the requirement of 10 CFR 50.68(b)(6) when fuel is present in these racks.

Please revise and update the US-APWR DCD sections 3.1, 9.1 and 12.3 to describe which radiation monitoring equipment is provided to meet the requirement of 10 CFR 50.68(b)(6) when fuel is present in the refueling cavity temporary fuel storage racks, or provide the specific alternative approaches used and the associated justification.

09.01.02-48

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 63 "Monitoring fuel and waste storage," requires systems to ensure fuel safety. 10 CFR 52.47(a)(2) and 10 CFR 52.47(a)(5) require the applicant to describe and fuel handling systems and the kinds and quantities of radioactive materials produced in the facility.

The applicant’s response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13013-P (R0) “Thermal-Hydraulic Analysis for US-APWR Containment Racks.” MUAP-13013-P (R0) Section 5.1 “Local Water Temperatures,” states that the water space is modeled as extending 6 feet above the containment racks. Institute of Nuclear Power Operations (INPO) Event Report Level 3 number 12-33 “Dislodged Steam Generator Nozzle Dam Bolting and Drain Plug During Eddy Current Inspections,” described a near rapid drain down event due to steam generator nozzle dams bolts that were missing causing the nozzle dam to bow out at the bottom because of the hydrostatic head pressure from the water in the refueling cavity. Should a nozzle dam fail, water level in the area of the temporary fuel racks could decrease to the level of the reactor vessel flange.

Please revise and update US-APWR MUAP-13013-P to utilize calculation assumptions that are consistent with relevant industry operating experience related to potential accident conditions, or provide the specific alternative approaches used and the associated justification.

09.01.02-49

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 63 "Monitoring fuel and waste storage," requires systems to ensure fuel safety.

The applicant's response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13012-P (R0) "Mechanical Analysis for US-APWR Containment Racks" MUAP-13012-P (R0) Section 4.1(2) "Straight Deep Drop Event" states that deep drop scenarios do not need to be analyzed for the containment racks since the containment racks do not have an elevated baseplate or rack pedestals. However, unlike the spent fuel pool, the dummy fuel element may be moved and lifted without being surrounded by water. Since MUAP-13012-P (R0) is silent with respect to dry handling and storage of the dummy bundle in the refueling cavity, it is not clear to the staff that the analysis bounds scenarios that have the potential to cause liner damage.

Please revise and update US-APWR MUAP-13012-P to utilize calculation assumptions that are consistent with relevant industry operating experience related to potential accident conditions, or provide the specific alternative approaches used and the associated justification.

09.01.02-50

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 "Criticality Safety of Fresh and Spent Fuel Storage and Handling," adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-41, dated 27 January 2012, the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-41, dated 25 April 2012, stated that the Containment Rack design is based upon unborated water with center-to-center spacing that would prevent criticality. Therefore, the introduction of unborated water to the fuel assemblies in the Containment Racks would not affect subcriticality, and as a result, Technical Specification (TS) 3.9.1 needs no modification. TS 3.9.1 is only applicable during MODE 6. MODE 6 is not applicable when all fuel is out of the reactor vessel. However, the applicant's response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) "Criticality Analysis for US-APWR Containment Racks." MUAP-13011-P (R0) Subsection 2.3.1.3.2 "Mislocated Fresh Fuel Assembly," states that "The mislocation of a fresh fuel assembly could, in the absence of soluble neutron

absorber, result in exceeding the regulatory limit (k_{eff} < 0.95). In addition, MUAP-13011-P (R0) Table 2-7 "Summary of CR Accident Case Calculations," shows that the Maximum k_{eff} (0 ppm soluble boron) is 1.0452.

Please revise and update the US-APWR DCD Technical Specifications section 3.9.1 to reflect the requirement for maintaining soluble boron reactivity controls consistent with 10 CFR 50.68(b)(4), when fuel is in the containment racks, and no fuel is in the reactor vessel, or provide the specific alternative approaches used and the associated justification.

09.01.02-51

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 "Criticality Safety of Fresh and Spent Fuel Storage and Handling," adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to adding Technical Specifications subsection 4.3.1.3, including the statement that the containment racks are designed and shall be maintained with k_{eff} less than 0.95 if fully flooded with unborated water. However, the applicant's response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) "Criticality Analysis for US-APWR Containment Racks." MUAP-13011-P (R0) Subsection 2.3.1.3.2 "Mislocated Fresh Fuel Assembly," states that "The mislocation of a fresh fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit (k_{eff} < 0.95). In addition, MUAP-13011-P (R0) Table 2-7 "Summary of CR Accident Case Calculations," shows that the Maximum k_{eff} (0 ppm soluble boron) is 1.0452.

Please revise and update the US-APWR DCD Technical Specifications to reflect the requirement for maintaining reactivity controls consistent with 10 CFR 50.68(b)(4), or provide the specific alternative approaches used and the associated justification.

09.01.02-52

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 "Criticality Safety of Fresh and Spent Fuel Storage and Handling," adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to changing subsection DCD 3.1.6.3.1 "Discussion," to stated that the containment racks are designed to have sufficient separation between adjacent fuel assemblies so the maximum k_{eff} under worst- case conditions is less than 1.0 without credit for the soluble boron, and less than 0.95 with partial credit taken for soluble boron. However, the applicant's response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) "Criticality Analysis for US-APWR Containment Racks." MUAP-13011-P (R0) Subsection 2.3.1.3.2 "Mislocated Fresh Fuel Assembly," states that "The mislocation of a fresh fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit ($k_{eff} < 0.95$). In addition, MUAP-13011-P (R0) Table 2-7 "Summary of CR Accident Case Calculations," shows that the Maximum k_{eff} (0 ppm soluble boron) is 1.0452.

Please revise and update the US-APWR DCD to reflect the requirement for maintaining reactivity controls consistent with 10 CFR 50.68(b)(4), or provide the specific alternative approaches used and the associated justification.

09.01.02-53

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 "Criticality Safety of Fresh and Spent Fuel Storage and Handling," adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-41 dated 27 January 2012 the staff asked the applicant to describe how the US-APWR DCD Technical Specification ensured compliance with the requirements of 10 CFR 50.68(b)(4) when fuel is in the containment racks. The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-41, dated 25 April 2012, contained statements such as "TS 3.9.2 on Unborated Water Source Isolation Valves is applicable in MODE 6 and would be applicable when the Containment Racks are in use as well," and "TS 3.9.3 on Nuclear Instrumentation is applicable in MODE 6 and applies without modification when fuel assemblies are temporarily stored in the Containment Racks." US-APWR DCD Revision 3 subsection 1.1. "Definitions," MODE which states that a MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. The response to RAI 895-6172 Revision 3 Question 12.03-12.03-41 did not commit to changing the definition of MODE 6, therefore, the noted response to RAI 895-6172 Revision 3 Question 12.03-12.03-41 is not consistent with the definition of MODE 6 when fuel is out of the reactor vessel, but present in the containment racks.

Please revise and update the US-APWR DCD Technical Specifications definition of MODE 6 or add a TS Applicability statement in the applicable TS (e.g. TS 3.7.10, 3.8.2, 3.8.5, 3.8.8, 3.8.10, 3.9.1, 3.9.2, 3.9.3, 3.9.4, and 3.9.7) to address "when one or more irradiated or new fuel assemblies are seated in the refueling cavity containment racks," that reflect the requirements for controls and storage of fuel consistent with 10 CFR 50.68(b), or provide the specific alternative approaches used and the associated justification.

09.01.02-54

As noted within SRP 9.1.5, "Overhead Heavy Load Handling Systems," Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 4 applies to SRP Section 9.1.5 because GDC 4 specifies protection against the effects of internally-generated missiles (i.e., dropped loads). A dropped heavy load in a critical area could cause a release of radioactive materials, a criticality accident, or inability to cool fuel.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the how the design described in US-APWR DCD provided adequate protection for fuel located in the containment racks. The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, contained committed to changing section 3.1 to include a description of other DCD sections that further discussed fuel handling and storage systems inspection and testing, decay heat removal, purification, and prevention of reduction in coolant storage inventory, which included section 9.1.5 "Overhead and Heavy Load Handling System." The guidance contained in SRP section 9.1.5 "Overhead Heavy Load Handling Systems," states that Safe load paths should be defined for movement of heavy loads to minimize the potential for a load drop on irradiated fuel and that paths should be defined clearly in equipment layout drawings. However, US-APWR DCD Revision 3 Figure 9.1.5-4 "Traveling Route of Heavy Load inside Containment," does not indicate the location of the containment racks, which may contain irradiated fuel.

Please revise and update the US-APWR DCD to indicate the locations where irradiated fuel may be stored within the refueling cavity (i.e. the containment racks), or provide the specific alternative approaches used and the associated justification.

09.01.02-55

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to provide additional information about the temporary fuel storage racks located in the Refueling Cavity. The applicant's response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, stated that in the event that the refueling cavity low-level water alarm became inoperable for any reason that the spent fuel pit water level alarm would be used to alert operators to take action if a leak occurred while fuel was in the containment racks. Proposed changes to the US-APWR DCD section 9.1.4.2.1.13 states that that in the event that the refueling cavity water level alarm RCS-LIA-01 1-N becomes inoperable, the spent fuel pit water level alarm SFS-LIA-010-N and SFS-LIA-020-N will be utilized. Proposed section 9.1.4.2.2.2 states that the low water level alarm of the refueling cavity is set at the required water depth for providing radiation shielding described in US-APWR DCD Subsection

12.3.2.2.4, and that the level meter of the SFP acts as alternative measurement of the refueling cavity level during the transfer of fuel. However, the methodology for establishing the setpoints of SFS-LIA-010-N and SFS-LIA-020-N, and how those setpoints are conservative with respect to ensuring that loss of shielding of irradiated components is detected in sufficient time to allow operators to take effective corrective actions, are not described in US-APWR DCD Revision 3 Chapters 5 or 9, so it is not clear that the setpoint of these monitors are conservative with respect to the Refueling Cavity level monitor setpoint requirements. See US-APWR DCD RAI 1033-7090 Probabilistic Risk Assessment and Severe Accident Evaluation dated May 13, 2013, for related questions.

Please revise and update the US-APWR DCD to describe the methods for ensuring that alternate level monitoring instruments are able to alert operators to maintain the required water depth for providing radiation shielding in the refueling cavity, or provide the specific alternative approaches used and the associated justification.

09.01.02-56

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 63 "Monitoring fuel and waste storage," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 64 "Monitoring radioactivity releases," requires systems to monitor for excessive radiation levels in the reactor containment atmosphere and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, and to initiate appropriate safety actions.

US-APWR DCD Revision 3, Section 3.1.6.4 Criterion 63 – “Monitoring Fuel and Waste Storage,” discusses area radiation monitoring and airborne radioactivity monitoring for fuel stored in the Spent Fuel Pool and new fuel storage racks, however, this section does not discuss how the US-APWR design addressed radiation monitoring for GDC 63 and GDC 64 for fuel stored in the containment racks.

Please revise and update the US-APWR DCD section 3.1.6.4 to describe the radiation monitoring instruments provided to meet GDC 63 and GDC 64 for fuel located in the containment racks, or provide the specific alternative approaches used and the associated justification.

09.01.02-57

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities", Part 68 “Criticality accident requirements,” paragraph (b)(6) states “Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.” Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 63 "Monitoring fuel and waste storage," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. GDC 64 "Monitoring radioactivity releases," requires systems to monitor for excessive radiation levels in the reactor containment atmosphere and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, and to initiate appropriate safety actions. 10 CFR 52.47(b)(1), which

requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and should operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

US-APWR DCD Revision 3 section 2.7.6.6 “Process Effluent Radiation Monitoring and Sampling System (PERMS),” does not describe the ITAAC for checking radiation monitoring equipment provided to meet the requirements of 10 CFR 50.68(b)(6). The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to adding section 2.7.6.14 “Containment Racks,” and Table 2.7.6.14-1 “Containment Racks Inspections, Tests, Analyses, and Acceptance Criteria.” However, these additions do not address the radiation monitoring equipment required by 10 CFR 50.68(b)(6), GDC 63 and GDC 64, for fuel stored in the containment racks.

Please revise and update the US-APWR DCD sections 2.7 or 2.7.6.6, to describe the ITAAC for radiation monitoring equipment provided to meet the requirement of 10 CFR 50.68(b)(6), GDC 63 and GDC 64 when fuel is present in the refueling cavity temporary fuel storage racks, or provide the specific alternative approaches used and the associated justification.

09.01.02-58

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to provide additional information about the temporary fuel storage racks located in the Refueling Cavity. The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, stated that in the event that the refueling cavity low-level water alarm became inoperable for any reason that the spent fuel pit water level alarm would be used to alert operators to take action if a leak occurred while fuel was in the containment racks. Proposed changes to the US-APWR DCD section 9.1.4.2.1.13 states that that in the event that the refueling cavity water level alarm RCS-LIA-01 1-N becomes inoperable, the spent fuel pit water level alarm SFS-LIA-010-N and SFS-LIA-020-N will be utilized. Proposed section 9.1.4.2.2.2 states that the low water level alarm of the refueling cavity is set at the required water depth for providing radiation shielding described in US-APWR DCD Subsection 12.3.2.2.4, and that the level meter of the SFP acts as alternative measurement of the refueling cavity level during the transfer of fuel. However, the changes committed to in the applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, did not discuss any requirement to ensure that the fuel transfer gate valve, or the spent fuel pool weir gate remain open, while fuel is in the refueling cavity racks, no fuel is in the reactor vessel, and no fuel movement is in progress.

Please revise and update the US-APWR DCD to describe plant configuration requirements for ensuring that alternate level monitoring instruments are able to alert operators to maintain the required water depth for providing radiation shielding in the refueling cavity, or provide the specific alternative approaches used and the associated justification.

09.01.02-59

Title 10 of the Code of Federal Regulations (10 CFR), Part 50 "Domestic Licensing of Production and Utilization Facilities" Appendix A "General Design Criteria for Nuclear Power Plants" (GDC) 61 "Fuel storage and handling and radioactivity control," requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. 10 CFR 50.68(b)(4) requires that if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, if flooded with unborated water. SRP 9.1.1 “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” adds clarification by stating that when fully loaded and flooded with full-density unborated water, the K(eff) will not exceed 0.95 for all normal and credible abnormal conditions.

In RAI 895-6172 Question 12.03-12.04-40 dated 27 January 2012 the staff asked the applicant to describe how the containment fuel racks comply with the requirements of 10 CFR 50.68(b)(4). The applicant’s response to RAI 895-6172 Revision 3 Question 12.03-12.03-40, dated 25 April 2012, committed to issuing three new Technical Reports describing the Containment Rack design basis. The applicant’s response to RAI 906-6332 Question 09.01.02-26, dated May 23 2013, included Technical Report MUAP-13011-P (R0) “Criticality Analysis for US-APWR Containment Racks.” However, MUAP-13011-P (R0) subsection 2.3.1.3 “Abnormal Location of a Fuel Assembly,” subsection 2.3.1 “Abnormal and Accident Conditions,” and MUAP-13011-P (R0) Figure 2-4 “MCNP Model for Fuel Displacement with Cells or CR,” do not provide a description of the limiting locations of fuel located in the racks and the dropped bundle.

Please revise and update Technical Report MUAP-13011-P to describe the limiting arrangement of fuel used for performing the analysis for ensuring reactivity controls are maintained consistent with 10 CFR 50.68(b)(4) when analyzing for a dropped fuel bundle with fuel in the containment racks, or provide the specific alternative approaches used and the associated justification.

09.01.02-60

The applicant’s response to RAI 895-6172, Question 12.03-12.04-41, dated April 24, 2012, provided additional information regarding Technical Specifications (TS) for the use of Containment Racks (CR) as temporary storage locations. Specifically, the applicant stated TS 3.9.5, Residual Heat Removal (RHR), is not necessary in order to cool the six assemblies located in the CR because the buildup of decay heat is not sufficient to require RHR cooling. However, the applicant has not provided an adequate heat up analysis that demonstrates the assemblies located in the refueling cavity, and the bulk refueling cavity temperature, will remain below acceptable temperatures without RHR cooling.

Compliance with GDC 61 requires, in part, that fuel storage be designed to ensure adequate safety under normal and postulated accident conditions, this includes suitable shielding for radiation protection, residual heat removal, and the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

The staff requests the applicant to provide an analysis that demonstrates all fuel assemblies located in the refueling cavity will remain protected under all operating conditions. RG 1.13, “Spent Fuel Storage Facility Design Basis,” provides guidance for protection of stored fuel; this includes temporarily stored fuel in the CR. The analysis should be consistent with the methodologies of RG 1.13 and consider the following:

a. The maximum expected number of potential assemblies (e.g. six in the CR, and one in transfer or in the upender).

b. Since a seismic category I system (RHR) is not credited for cooling the refueling cavity for all operational conditions, demonstrate that the bulk refueling cavity temperature remains below acceptable limits under normal and abnormal conditions (justify limits). The refueling cavity should have a seismic category I makeup water system and backup makeup system with sufficient makeup capacity higher than the worse boiloff rate. Specify the seismic category I water source credited for maintaining cavity water level above TS 3.9.7 limits. Also, spent fuel pool pumps should not be credited unless there is a TS requirement to have the fuel transfer tube valve/gate open when there are fuel assemblies in the refueling cavity.

c. The analysis should calculate the time to exceed bulk refueling cavity design temperature without RHR cooling in both normal operation (i.e. cavity water level at TS minimum), and accident conditions (e.g. failure of a non-seismic component, such as a nozzle dam, resulting in refueling cavity water level to drain to the reactor vessel flange) to ensure plant personnel have adequate time to respond to the event. Justification is necessary for an assumed starting water temperature above 140^0F.

d. Discuss how the containment building and its ventilation and filtration system will handle the boiling water of the refueling cavity when in no mode (i.e. no fuel in the vessel, and up to seven assemblies in the refueling cavity).

In addition, DCD Revision 3, Section 5.4.7.2.3.5 states, "During refueling, the RHRS is maintained in-service with the number of pumps and heat exchangers in operations as required by the heat load." This statement is vague and it is unclear to the staff what heat loads would require RHRS to be in-service. This statement and any other affected DCD sections should be revised or updated based on the analysis discussed above regarding the protection of fuel assemblies located in the refueling cavity.

09.01.02-61

Follow-up to Thermal-Hydraulic Analysis for US-APWR Containment Racks

On May 21, 2013, the staff received a technical report titled, "Thermal-Hydraulic Analysis for US-APWR Containment Racks." This report concluded no local boiling occurs anywhere along the fuel rods in the Containment Racks (CR), and the peak fuel cladding temperatures for all of the stored assemblies remains below the local saturation temperature of the surrounding water. However, the staff cannot make the same conclusion because some of the assumptions in the analysis are missing or are not adequately justified.

The staff requests the applicant to provide additional information regarding the following items:

a. On page 5, the analysis states that the radial peaking factor is taken into consideration as part of the maximum decay heat per fuel rod (Qrod), however, this value is missing from Table 6-1. What is the maximum decay heat per fuel rod without radial or axial peaking factors, and what are the assumptions for that value (i.e. how many hours after shutdown is assumed before the assembly is moved to the CR).

b. The analysis uses an initial bulk water temperature of 160^0F. However, this temperature is inconsistent with RG 1.13, "Spent Fuel Storage Facility Design Basis," which states bulk water temperature for stored fuel should remain below 140^0F, and DCD Revision 3, Section 3.8.1.5.3, which states for normal operation concrete temperatures are not to exceed 150^0F. It is unclear to the staff if the use of 160^0F in the analysis is for conservatism or is an actual expected operating temperature. If the 160^0F is only for conservatism, what is the maximum normal operating bulk water temperature of the refueling cavity?
c. On page 11, the analysis states the minimum depth of water at the top of active fuel length is approximately 29 feet. It is not clear to the staff whether this depth is calculated from normal water level or the minimum required water level by TS 3.9.7 on Refueling Cavity Water Level (23 feet above the vessel flange). Justify the use of a minimum water depth of 29 feet and why it is conservative.

09.01.02-62

In MUAP-13012-P (R0), Section 3.1, "Modeling Methodology," (Page 3) the paragraph states, in part, "The acceleration corresponding to the lowest natural frequency is used as the inertial amplifier to realize the seismic load effects."

The applicant is requested to address the following questions:

1. Provide a description of the mathematical model of the containment rack that is used to compute the natural frequencies.

2. The lowest natural frequency may be that of a local mode not a structural mode; therefore, the mode corresponding to the lowest natural frequency may not be the dominant mode. Therefore, provide data to show that the mode corresponding to the lowest natural frequency is the dominant structural mode.

3. Provide data to show that the contributions from higher modes are negligible.

The above information should be included in the applicable portion of the technical report.

09.01.02-63

In MUAP-13012-P (R0), Section 3.3, "Assumptions," (Page 6) the item (1) states that "Conservatively, the fluid coupling effects surrounding the racks are neglected. The absence of fluid effects increases the rattling of the rack thereby overpredicting the stresses (or deflection) in the rack."

The fluid coupling effects surrounding the racks should not be neglected because it may lower the natural frequency of the rack and increase the forces exerted at the rack supports. Therefore, the applicant is requested to provide technical reasons to demonstrate that neglecting the fluid coupling effect between the racks and water would be conservative for the rack design. The above information should be included in the applicable portion of the technical report.

09.01.02-64

In MUAP-13012-P (R0), Section 3.3, "Assumptions" (Page 6), item (2) states that "While evaluating the natural frequency of the rack, the fluid mass is considered uniformly distributed over the entire height of the rack cells thereby underestimating the natural frequency. The lower natural frequency results in larger corresponding acceleration used as seismic amplifier in the current evaluation."

The applicant is requested to address the following questions:

1. Provide the amount of fluid mass that was included in the frequency calculation.
2. The fluid mass represents only the impulsive part of liquid motion against the rack during earthquakes. Was the liquid sloshing motion against the rack considered in the analysis?
3. Provide the water depth and the dimensions of the refueling cavity.
4. The lower natural frequency may not result in larger corresponding acceleration. It depends on the shape of floor response spectra. Provide information for the floor response spectra used and the natural frequencies with and without the fluid for the rack.

The above information should be included in the applicable portion of the technical report.

09.01.02-65

In MUAP-13012-P (R0), Section 3.7.1.1 "Corresponding Acceleration," the equation given for calculating the natural frequency of the rack is (Page 8)

f_n = (22.4/2 π)*Square root((E*I*g)/(w_1*L_1^4))

Where:

w_1= Load per unit length of unsupported cells (27.21 lbf/in)

The applicant is requested to provide detailed information as to how the 27.21 lbf/in is derived and whether or not w_1 includes the fluid mass.

The above information should be included in the RAI response.

09.01.02-66

In MUAP-13012-P (R0), Section 3.7.1.1 "Corresponding Acceleration," the third and fourth paragraphs state that "The horizontal acceleration is the maximum acceleration in X-direction or Y-direction. The Floor Response Spectra (FRS) is shown in Reference 6-8. As a result, horizontal acceleration is 0.78g and vertical acceleration is 1.33g."; and "It has been confirmed that the FRS (Reference 6-8) used in this report are larger than the one provided in Reference 6-10."

The applicant is requested to address the following questions:

1. Does "The horizontal acceleration is the maximum acceleration in X-direction or Y-direction," mean that the applicant only considered the larger magnitude of the acceleration of the two horizontal accelerations in the X and Y directions for the analysis of the rack? If that is the case, how does the applicant address the combined effects for the rack design due to earthquakes in the X, Y, Z components?
2. Since the staff does not have access to Reference 6.8, provide the floor response spectra for X, Y, and Z directions with descriptions including their elevations used for the rack analyses, and describe how 0.78g and 1.33g are obtained.
3. How is the natural frequency of the rack in the vertical direction calculated?
4. Explain the meaning and purpose of the statement "It has been confirmed that the FRS (Reference 6-8) used in this report are larger than the one provided in Reference 6-10."

The above information should be included in the applicable portion of the technical report.

09.01.02-67

In MUAP-13012-P (R0), Section 3.7.1.2, "Fuel-to-Cell Wall Impact Loads," the first paragraph (Page 8) states that "To evaluate the impact loads between the fuel assembly and cell wall, a 2-D model of the CR plus three stored fuel assemblies has been developed using the computer code MR216 (also known as DYNARACK). Each stored fuel assembly is modeled as a lumped mass with 2 translational degrees of freedom in the horizontal x and y directions. The CRs are modeled as a separate body with 3 degrees of freedom (2 translational, 1 rotational). Each fuel mass is surrounded by 4 gap elements (in the +x, +y, -x, and -y directions), which track the impacts between the fuel and the CR during the seismic event. This model includes the effects of fluid coupling between the fuel assemblies and the cell walls."

The applicant is requested to address the following questions:

1. Provide description for the mathematical model of the rack and its boundary conditions and the earthquake input motions to the model.
2. Is the performed impact analysis a linear or nonlinear structural analysis? If it is a linear analysis, the applicant is requested to provide technical details for how this analysis is carried out; if it is a nonlinear analysis, according to SRP 3.7.1, more than four time histories should be used. Provide and describe the number of time histories used in the analysis.
3. Provide a sketch for the 2-D model and describe how the fuel mass is calculated.
4. When subjected to the horizontal seismic motion, the deflection of the containment rack is not uniform along its height because the containment rack is anchored to the adjacent wall only at its top and bottom. Also due to the effect of soil-structure interaction, the reactor complex experiences a rocking motion at its base; as a result of this, the motion at the base of the containment rack is different from that at the top of the rack. A 2-D model in x-y plane of the containment rack may not be appropriate. Provide technical rationale to justify that the 2-D model is appropriate, and why a 3-D model is not required.

The above information should be included in the applicable portion of the technical report.

09.01.02-68

In MUAP-13012-P (R0)), Section 3.7.1.2, "Fuel-to-Cell Wall Impact Loads", the second paragraph (Page 8) states that "The maximum fuel-to-cell impact load calculated by this model is 4,628 lbf shown in Table 3-4 (Reference 6-11). The maximum allowable impact load on the cell membrane, considering an additional safety factor of 2, based on the yield strength is then given as 24,994 lbf."

The applicant is requested to address the following questions:

1. How is the allowable impact load of 24,994 lbf on the cell membrane obtained?
2. What is the allowable impact load for the fuel assembly? The applicant needs to show that the fuel assembly is not damaged by the impact loads.

The above information should be included in the RAI response.

09.01.02-69

In MUAP-13012-P (R0), Section 3.7.2, "Rack Stress Evaluation," the paragraph (Page 9) states that "Regarding supports of the fuel cell (see Figure 2-1), the net section maximum bending moments, shear forces and tensile stress can also be determined at the mounting angle. Based on these, the maximum stress in the CR can be evaluated at the mounting angle. The evaluation of the fuel cell and mounting angle is described below. In addition, the results at other points, such as the end rigger plates, are shown in Tables 3-5 and 3-6."

The size and dimensions of the mounting angle and rigger plates are not given in Figure 2-1 or in Tables 3-5 and 3-6. The applicant is requested to provide this information. The above information should be included in the applicable portion of the technical report.

09.01.02-70

In MUAP-13012-P (R0), Section 3.7.2.1, "Fuel Cell," the bending stress is given by the equation on page 9:

Bending stress = $(M_1/Z_1) + (M_2/Z_2) = 1761$ psi

Where:

Z_1 : X-axis section modulus of the rack (211.9 in^3)
Z_2 : Y-axis section modulus of the rack (95.95 in^3)

The applicant is requested to address the following questions:

1. Describe how M_1 and M_2 are calculated.
2. Provide dimensions/sketches of the cross section of the fuel cell.
3. Why are Z_1 and Z_2 the moduli of the rack and not moduli of the fuel cell since the calculation pertains to the fuel cell?

The above information should be included in the applicable portion of the technical report.

09.01.02-71

In MUAP-13012-P (R0), Section 3.7.2.2, "Mounting Angle Tensile and Bending," (Pages 9 and 10) the paragraph states the following:

W_2 : Reaction at limiting cross section (6,168 lbf)
A_1 : Area of the mounting angle limiting cross section (13.62 in^2)
M_3 : Moment about VT-axis (44,440 lbf in)
M_4 : Moment about Y-axis (58,160 lbf in)
Z_3 : VT-axis section modulus of the rack (2.047 in^3)
Z_4 : Y-axis section modulus of the rack (44.31 in^3)

The applicant is requested to address the following questions:

1. Provide details/sketches to show how these quantities are calculated.
2. Both Z_4 in Section 3.7.2.2 and Z_2 in Subsection 3.7.2.1 represent the Y-axis section modulus of the rack. Why Z_4 has a value of 44.31 in^3, which is different from Z_2 that has a value of 95.95 in^3?

The requested information should be included in the applicable portion of the technical report.

09.01.02-72

In MUAP-13012-P (R0), Section 3.7.2.3, "Mounting Angle Shear Stress," (Page 10) the first paragraph states that "The moment about X-axis is resisted by a force couple on the pair of mounting angles on opposite sides of the CR. This force couple causes a shear stress to develop in the mounting angles. This shear stress is combined with the shear stress caused by the load in the Y direction and the VT-direction acceleration in the following calculation."

The applicant is requested to provide a sketch that shows how the resulting shear was formed for the CR and how the resisting shear was provided by the mounting angle. The requested information should be included in the RAI response.

09.01.02-73

In MUAP-13012-P (R0), Section 3.7.2.4, "Mounting Angle-to-Wall Welds," (Page 11) the paragraph states that "The angle plates may be welded directly to the wall by 3/16 inches fillet welds. The stress in the anchor weld is calculated below."

Stress in anchor = W_4/A_3 = 1052 psi

Where:

W_4 : Combined shear force on one mounting angle (4,738 lbf)
A_3 : Area of weld between mounting angle and wall (4.508 in^2)

The applicant is requested to provide details/sketches as to how W_4 and A_3 are calculated and the type of weld material used in fillet weld. The requested information should be included in the RAI response.

09.01.02-74

In MUAP-13012-P (R0), Section 4.1 (1), "Straight shallow drop event," (Page 12) the paragraph states that "In the so-called "straight shallow drop" event, an impactor (i.e., a fuel assembly plus its rod cluster control assembly) is assumed to drop vertically and hit the top of the rack."

The applicant is requested to explain whether the weight of the fuel handling tool is included in the impactor. If not, why? The requested information should be included in the RAI response.

09.01.02-75

In MUAP-13012-P (R0), Section 4.3.1, "Straight Shallow Drop Event" (Page 15), the applicant states that the straight shallow drop event does not compromise the structural integrity of the CR. The applicant, however, does not address the structural integrity of the fuel assembly.

The applicant is requested to address the structural integrity of the fuel assembly. Also, the applicant is requested to provide information on how the design limit of 6.5 inches for the CR cellular structure plastic deformation and the allowable weld stress of 35,690 psi mentioned in this section are obtained. The above information should be included in the applicable portion of the technical report.

09.01.02-76

In MUAP-13012-P (R0), Section 4.3.2, "Stuck Fuel Event," (Page 16) the paragraph states that "Result of the analysis show that the maximum stress in the rack cell due to a stuck fuel assembly is only 1,197 psi, which is well below the material yield stress. Therefore, the CR are adequate to withstand a 4,400 lbf uplift force due to a stuck fuel assembly."

The applicant is requested to provide the following information:

1. A description of the mathematical model and assumptions used in this analysis.
2. How is the 4,400 lbf uplift force determined to be enough to uplift the stuck fuel?
3. The location of the maximum stress of 1,197 psi. Does the location of the maximum stress depend on the assumption of how the fuel assembly gets stuck?
4. Are the mounting angles and welds adequate for the 4,400 lbf uplifting force?
5. What is the maximum stress developed in the fuel assembly? The applicant is requested to show that the integrity of the fuel assembly is maintained.

The above information should be included in the applicable portion of the technical report.

09.01.02-77

In DCD Revision 3, Mark-up Section 9.1.2.2.4, "New Fuel Storage Rack, Spent Fuel Storage Rack and Containment Rack Design," (Page 9.1-11) the first paragraph states, "Structural design and stress analysis of the new fuel storage racks, spent fuel storage racks and containment racks are evaluated in accordance with the seismic Category I requirements of Regulatory Guide 1.29."

The staff noticed that Regulatory Guide 1.29, "Seismic Design Classification," does not contain the structural design and stress analysis requirements for seismic Category I structures. The applicant is requested to clarify that the reference RG 1.29 is for classification and to provide the evaluation criteria for the new fuel racks, the spent fuel racks, and the containment racks. The applicant response should be included in the applicable portion of the DCD.